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sists of 61 pages.

July 16, 1969

DOUGLAS UNITED NUCLEAR

MONTHLY REPORT

JUNE 1969

DOUGLAS UNITED NUCLEAR, INC.

Richland, Washington

Work performed under Contract No. AT(45-1)-1857 between the
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SUMMARYREACTOR PLANT OPERATIONSProduction Statistics

	<u>KE</u>	<u>KW</u>	<u>N</u>
Input Production - Pu (KMWD)	106.6	86.9	66.5
- U-233 (Equiv. KMWD)	17.0	12.8	-
Time Operated Efficiency - %	91.1	76.7	59.6
Steam Availability to WPPSS - %	-	-	57.9

K Reactors

Maximum power levels of 3940 MW at KE and 3950 MW at KW were achieved while operating under restrictions imposed because of inlet water piping brittle fracture considerations.

At both reactors thoria core loads are being irradiated to make U-233; the plutonium produced was preponderantly fuel grade.

N Reactor

Maximum power level was 4000 MW, the administrative limit. Eight unscheduled outages occurred: a manual shutdown to repair leaks in steam generator 5B; six automatic scrams, two due to low-flow trips of the circulating raw water system, one due to trip of high average power level monitor, two due to low-flow trips on the process tube flow monitor, and one due to main steam header high pressure trip; and one manual scram following confirmation of a fuel failure (Mark IV).

Foster Wheeler boiler tube failures and repair extended the first reactor outage by about five days, and caused this boiler to be out of service again at month end.

Production Test-115, "Establishing CE Boiler Performance to Permit Reactor Startup Using CE Boiler Steam Supply", was completed satisfactorily.

FUEL AND TARGET FABRICATIONProduction Statistics (tons)

	<u>For KE & KW</u>	<u>For N</u>
Billets Extruded	-	55.4
Finished Fuel Produced	145.7	32.3
Thoria Canned	25.3	-

K Fuels & Targets

AlSi canning operations continued on the basis of two lines per day, five days per week. Work progressed on the relocation and modification of hot-die-sizing facilities under project DCE-524. All thoria canned was in wafer form.

N Fuels

Output production was again slightly below forecast because of manpower diversion to N Reactor to assist with the inspection of fuels potentially damaged in charging.

TECHNICAL ACTIVITIESK Reactors

Fracture mechanics analyses and tube power limits for the K reactors have been summarized in a supplementary Safety Analysis Report (HW-74095 SUPP5) and discussed with the Division of Reactor Licensing, AEC, late in June. Major conclusions of the report are that (1) adequate technology has been developed for establishing limits insuring against fission product release in event of major pipe failure, (2) it is desirable that certain high-stress piping joints be replaced, and (3) the piping system integrity can be adequately determined by periodic proof testing.

Out-of-reactor thermal cycling tests on irradiated Zircaloy process tubing have indicated that temperature cycling, as occurs during reactor startup and shutdown, is not the driving force for moving hydride from the case layer into the base metal.

Initial examination was made of two Zircaloy tubes from KE Reactor which had been grit-blasted on September 10, 1966, and subsequently operated with anodized dummies. The examination agrees with earlier observations that no further hydriding occurs where the case layer removal by grit-blasting has been complete.

The single column of americium targets charged into KE on May 27 to obtain information on the production of Pu-238 has operated satisfactorily. Five test columns to demonstrate the production of medical grade Pu-238 from Np-237 were charged into KW on June 16.

N Reactor

Evaluation of the Mark IV demonstration test fuel performance data is about 50 percent complete. The swelling rate on the Mark IV fuel monitored to date has been linear with exposure in the range observed. Remaining Mark IV monitor columns are goaled to be irradiated to high exposures for comparison with Mark I high exposure performance data.

Hydrogen analyses from a process tube removed in February indicate that hydrogen buildup is not continuing with operation, whereas a positive buildup trend

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could have been inferred by comparison of data from process tubes removed in 1966 and 1967.

Operation of the 324 Building D cell was initiated on June 5 with apparently successful tests using a high-exposure short-cooled fuel element. Measurements of metal-water reaction rates and fission product release rates were obtained for an irradiated Zircaloy-clad fuel heated to just below the melting temperature of the uranium.

Engineering study of the V-12 diversion valves continued, with emphasis on the selection of corrosion resistant materials for the actuator. In the improvement of ball safety system reliability, two prototype ball hopper trip mechanisms were assembled for testing and evaluation. Based on preliminary trials, development effort is being concentrated on the direct-coupled design.

FEATURE REPORT

The appended topical report summarizes the theory of fracture mechanics analysis, and the reassuring results obtained in the application of this analysis to an assessment of brittle fracture potential in the K reactor inlet piping.

GENERAL

Two Negro Co-op students were employed on June 9. Both have been accepted at Washington State University for entry this fall. The Company now has four minority employees participating in this Cooperative Education Program.

The force reduction resulting from C Reactor deactivation is continuing. Contract personnel at month end totaled 1,761, a drop of 37 in June.

There were no disabling injuries and no radiation exposures exceeded operational control.

for Charles D. Harrington
President

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REACTOR PLANT OPERATIONS - KE & KW

PRODUCTION

General

Production, power levels, efficiencies and related statistics for the two K reactors are tabulated below. Overall K reactor input production and time operated efficiency (TOE) for the past six months are shown on the following chart:

Thoria loads are being irradiated in both KE and KW to make U-233. The plutonium produced was preponderantly fuel grade in the 8-10 percent Pu-240 range.

Statistical Summary

	KE	KW	Total
Input Production - Pu (KMWD)	106.6	86.9	193.5
- U-233 (Equiv. MWD)	16,958	12,777	29,735
Power Level (MW) - Maximum	3940	3950	7890
- Average	3902	3777	7679
Time Operated Efficiency - %	91.1	76.7	83.9
Number of Outages	1	3	4
Number of Startup Interruptions	0	3	3
Operating Coolant Flow - M gpm	190.0	191.3	381.3
Fuel Charge (Tons) - 94 Metal	356.9	357.4	714.3
- Natural U	3.0	5.5	8.5

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	<u>KE</u>	<u>KW</u>	<u>Total</u>
Fuel Element Failures	0	0	0
Helium Losses - M cu. ft.	116.7	133.5	250.2

OPERATING EXPERIENCE

Reactor Loadings

Front face maps showing the loadings in KE and KW are reproduced on the two pages which follow page B-4. The tonnages listed are approximate; actual fuel totals are tabulated above.

Power Levels

Maximum power levels shown for the K reactors represent continuing operation under precautions taken because of inlet water piping brittle fracture considerations.

Reactor Outages

<u>Date Down</u>	<u>Reactor</u>	<u>Outage Hours</u>	<u>Remarks</u>
June 4	KW	45.0	Reactor scram caused by an undervoltage relay trip on No. 6 low-lift pump during a Maintenance Standards check.
June 13	KW	94.5	Scheduled outage for charge-discharge and maintenance.
June 18	KE	64.3	Reactor shut down when No. 24 VSR dropped from its latching mechanism during operation. A small charge-discharge and some maintenance work were completed.
June 29	KW	28.3	Reactor scram caused by an unexplained Panellit trip.

EQUIPMENT EXPERIENCE

Flexible VSRs

Investigation of the No. 24 VSR drop at KE showed that several taper pins holding the rod sections together were loose and had caused binding in the step plug. Apparently the rod bound in the step plug, did not thoroughly seat in the holding latch, and subsequently slipped loose during reactor operation. A program is underway at both reactors to check all knuckle rods for tightness of taper pin installation.

Boiler Flameout - KE Reactor

Following an acceleration test on No. 1 turbogenerator at KE on June 4, the load was being reduced when the No. 1 boiler right-hand flame went out. Shortly following this, the left flame went out. The fire was relit within the time specified in Process Standards. A subsequent check revealed that mercury had blown from the fuel oil pressure sensing instrument, causing a poor mix of fuel oil and air. There was no cause found for the mercury having blown. Repairs were made and the system has been working well since the flameout.

Deactivation - C Plant

Deactivation work has progressed steadily since the reactor was shut down on April 25. It is estimated that Processing's work in the reactor building is 90 percent complete. Steam and water facilities work is about 56 percent complete.

PROCESS ASSISTANCE AND CONTROLOperational Physics

The transition from powdered thorium targets to wafered thorium targets had a negligible reactivity effect. Slight variations in target density and, therefore, in reactivity (0.1 percent $\Delta k/k$) are normally expected from cycle to cycle anyway.

The 13 PuAl targets in the KE Reactor are at intended powers, and thorium in the buffer zone will be continually replaced with ^{94}Pu Metal as the levels of Pu-239 and Pu-241 (fissionable isotopes) are reduced in the plutonium burning process. The 225-tube target-buffer zone block is causing no operational problems.

The ^{94}Pu Metal E-Q loads are operating as intended. The relatively high positive graphite temperature coefficient of reactivity (nearly double that for a natural uranium lattice) accentuates the xenon oscillation potential, but careful surveillance of operating parameter trends has precluded operational difficulties in this area.

Charge-discharge scheduling has taken full advantage of the xenon-135 poisoning following shutdown for total control purposes. Both the ^{94}Pu Metal and the thorium target material (which builds up U-233, a fissionable isotope) require increased total control compensation with increasing exposure, and timely discharge of certain materials from specific locations in the reactor can be beneficial for total control requirements.

Some operational physics parameters of interest are shown in the following table:

	Reactor	
	KE	KW
Effective Central Tubes (ECT)*	2256	2222

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	Reactor	
	KE	KW
Flattening Efficiency** - June	0.72	0.71
- 12-mo. average	0.72	0.71
Maximum Operating Time Permitting Scram Recovery - Hours***	10	11

*Reactor power level divided by the average power of the ten most productive tubes which are representative of the reactor loading.

**ECT divided by the number of power generating tubes.

***The maximum operating time subsequent to a cold startup following which a scram recovery could be made using the currently approved startup procedures.

Product Accountability Tables

Tables for the prediction of U-236 and Np-237 buildup have been revised slightly prior to documentation. To avoid confusion in the definitions of "gratuitous" and "incremental" neptunium between economic and technical studies, the table column headings are identified by neptunium source (whether from U-236 or from U-235 and U-238 reactions) permitting the user to utilize the tables flexibly.

Production Fuel Performance

The following table shows production fuel failure frequencies in the K reactors, as number per million elements discharged, for the 12- and 24-month periods ending May 20:

	<u>12-Months</u>	<u>24-Months</u>
Natural Uranium	7	7
94 Metal	3	7

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<u>Zone</u>	<u>Tons</u>	<u>Material</u>
Central	237	94 Metal (for Thoria Support)
	2	Natural Uranium
	19	Thoria ("X" Designates Tubes)
	1	Special Depleted Uranium (PITA-048)
Buckled	65	94 Metal (for Thoria Support)
	6	Thoria ("X" Designates Tubes)
Blanket	55	94 Metal (for Thoria Support)
	11	Thoria

Loading Pattern - KE Reactor

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<u>Zone</u>	<u>Tons</u>	<u>Material</u>
Central	237	94 Metal (for Thoria Support)
	1	Natural Uranium
	18	Thoria ("X" Designates Tubes)
	1	Special Depleted Uranium (PITA-048)
Buckled	66	94 Metal (for Thoria Support)
	6	Thoria ("X" Designates Tubes)
Blanket	53	94 Metal (for Thoria Support)
	11	Thoria

Loading Pattern - KW Reactor

B-B

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DUPLICATE

REACTOR PLANT OPERATIONS - N

PRODUCTION

General

Reactor production (all fuel grade Pu), power level, and related statistics are tabulated below. Input production and time operated efficiency (TOE) for the past six months are shown on the following chart:

Statistical summary

Input Production - Pu (KMWD)	66.48
Electrical Generation - KMW	317.29 (per WPPSS)
Power Level (MW) - Maximum	4000
- Average	3717
Time Operated Efficiency - %	59.6
Steam Availability - %	57.9
Number of Shutdowns - Scheduled	0
- Unscheduled	8
Fuel Failures	1
Fuel Charge (Tons) - 94 Metal	307.6
- 125 Metal	68.3
- Natural U	0.4
Total	376.3

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Helium Losses - M cu.ft.

409.578

Fuel Oil Usage - bbl.

18,160

OPERATING EXPERIENCE

Reactor Loading

The reactor loading at month end is shown on the front face map which follows page BN-5. Charge-discharge for the month consisted of the charging of two columns of Mark IV fuel (columns 1443 and 1843) under PT-NR-94, "Mark IV Fuel Demonstration". Recharging of the fuel failure tube 1167 will be reflected in the July report.

Power Level

Power level was administratively limited at 4000 MW. It was possible to maintain main steam header pressure (MSHP) at approximately 123 psig, thus enabling WPPSS to generate power at their design rating of 800 MWe.

Reactor Outages

The reactor was operating at maximum authorized power at the beginning of the month, and was in shutdown status at month end. The eight reactor outages and their principal causes were as follows:

<u>Date Down</u>	<u>Outage Hours</u>	<u>Cause</u>
June 8	183.9	Unscheduled shutdown to repair leaks in steam generator 5B. Outage was extended due to boiler tube failures during boiler startup.
June 15	34.2	Unscheduled scram due to trip-off of No. 1 circulating raw water pump because of high amperage indications.
June 17	2.0	Unscheduled scram from an unexplained low-flow trip on the circulating raw water system.
June 17	1.0	Unscheduled scram due to a trip on the high-level nuclear instrumentation. Scram occurred during change-out of a failed power supply.
June 19	21.9	Unscheduled manual scram because of the indicated loss of seal water to No. 1 primary pump. A failed sensing line had resulted in loss of seal water flow indication.
June 21	25.6	Unscheduled scram due to a main steam header pressure trip when the power supply to CR-39 was inadvertently lost.

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<u>Date Down</u>	<u>Outage Hours</u>	<u>Cause</u>
June 23	3.1	Unscheduled scram due to a low-flow monitor trip when No. 3 drive turbine (primary pump) tripped-off due to loss of vacuum.
June 30	19.0	Unscheduled manual scram upon confirmation of a fuel failure in process tube 1167.

Fuel Failure

Inspection of the Mark IV fuel failure which occurred in process tube 1167 on June 30 revealed a fretting condition on the inner bore of the outer assembly adjacent to a solid support. The failed element was the fourth piece from the rear of a fuel column charged last month.

EQUIPMENT EXPERIENCE

Primary Loop

When the reactor was shut down on June 8 for repair of tube leaks in steam generator 5B, the leak rate had increased to about 500 gallons per hour, and the ammonia in the secondary coolant had increased to greater than 2000 ppb. The broken tube and eight surrounding tubes were plugged and vented. At month end the leak rate in 5B had again increased to 33 gallons per hour. Repair of these leaks was initiated during the June 30 outage.

The No. 3 primary pump turbine trip-off on June 23 was due to low surface condenser vacuum which resulted from air in the steam header. The steam system was revented to correct the problem.

Eleven inlet connector valves were repaired during June. The primary loop leak rate dropped to about 155 gpm following the latest repair.

Re-tubing of the Cell 2 steam generators was completed during the month. Both steam generator heads were welded in place and stress relieved. Installation of auxiliary lines and piping insulation was underway at month end in preparation for final testing in July.

Boiler Experience

During the outage begun on June 8 a leak developed in the Foster Wheeler boiler. The boiler was shut down and a floor tube leak was repaired. During startup from this boiler outage, the steam drum was inadvertently allowed to drain too low and additional boiler tube failures resulted. The reactor outage was extended about 125 hours for boiler repair. Following this repair, the boiler was given a cold hydrostatic pressure test at 860 psig.

Production Test NR-115, "Establishing CE Boiler Performance to Permit Reactor Startup on CE Boiler Steam Supply", was completed satisfactorily. The test was run following the installation of new fire bricks in each boiler, and the correction of other defects noted during the overhaul.

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The Foster Wheeler boiler was again shut down for leaks on June 21. The CE boilers were used for reactor backup. Following leak repair, the FW boiler operated until June 26 when it was shut down for inspection by a Foster Wheeler engineer, and to repair additional leaks; it remained out of service through month end.

Circulating Raw Water System

The reactor outage on June 15 originated when amperage readings on No. 1 circulating raw water pump increased to 150 to 200 amperes during operation. The pump was manually tripped from the reactor Control Room, and the ensuing low CRW flow scrambled the reactor. Inspection of the pump motor revealed extensive damage to the slip rings. The cause of this damage was not determined. Repairs were effected by honing the slip rings, cleaning the motor, and installing new brushes.

A low differential pressure indication on CRW pump No. 3 may have been the cause of the CRW low-flow trip during startup on June 17. The Barton transmitter was recalibrated to a proper setting.

The raw water return valve on dump condenser No. 15 failed to close properly. Investigation revealed that the butterfly in the valve had broken and the shaft was badly bent. Cause of the failure is not known. The raw water supply valve was used to replace the return valve, so the dump condenser could be kept in service.

Secondary System

On June 21, electrical breaker CR-39 opened causing the main steam header pressure setpoint to fail to midscale. The main steam header pressure control system responded by closing the dump condenser inlet steam valves and the export steam valves, thus raising the main steam pressure to trip point. The breaker opened when a short to ground occurred while the miscellaneous temperature recorder was being repaired in the reactor Control Room. The breaker feeding the recorder also feeds the Bailey system GGG cabinet which in turn contains the main steam header pressure control circuitry. The recorder power source was transferred to a non-critical circuit to prevent recurrence.

Emergency Raw Water System

The emergency raw water tank was drained, cleaned and inspected during the outage initiated June 8. The emergency cooling water lines were then flushed with demineralized water and the system returned to normal. A leak developed in the relief valve standpipe at the V-19 drain valve station. The standpipe was replaced. Investigation revealed severe pitting in the pipe, including the one penetration. Additional inspection will be made prior to reactor piping decontamination to assure piping integrity and cleanliness. Thermocouples were installed downstream of the V-4 dump valves in preparation for V-4 leak-through inspection. Work was done in accordance with provisions of Development Test No. 55.

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PROCESS ASSISTANCE AND CONTROL

Operational Physics

The Mark IV fuel columns continue to operate just under the calculated power and reactivity, but within expected calculational tolerances. This fuel may therefore require 10-20 percent more spike columns than the Mark I fuel. This increase is offset by the recently demonstrated ability to control the reactor with a lower excess reactivity (compensated by rods) such that the size of the spike ring is not expected to change significantly due to the transition from Mark I to Mark IV fuel.

There were seven startups, including four of the "secondary cold" type. Special attention was required during startups to keep the axial flux distribution within prescribed limits. Use of the current inboard rod pattern can lead to some peaking in the front and rear. The rod pattern will be "spread out" somewhat to alleviate this problem in the future.

Some operational physics parameters of interest are:

Effective Central Tubes (ECT)	805
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Flattening Efficiency - June	0.80
- 11-mo. average	0.82

Maximum operating time permitting scram recovery - hours	24
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Product Accountability

Isotopic yield and burnout prediction tables for inclusion in the accountability manual have been completed and issued for Mark I and Mark IV fuels. Mark IV AA tables, covering the Mark IV design with 125 Metal in both the inner and outer components, are also being prepared.

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	41	42	43	44	45	46	47	48	49	50	51	52	53	54	55	56	57	58	59	60	61	62	63	64	65	66	67	68	69	70	71	72	73	74	
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Fuel Code	No. Tubes	Description	PT-NR No.	No. Tubes	Description
C	94	Mk-IC (94 Metal - Fringe)	76	4	Fuel Meltdown Test
CE	61	Mk-IC (94 Metal - Fringe)	94*	8	Mk-IV Demonstration
D	164	Mk-IC (94 Metal - High U-236)	96	16	Mk-1 From Upset-Forged Billets
E	219	Mk-IC (94 Metal - Fringe & Central)	01	12	Mk-1 From Direct-Cast Billets
F	112	Mk-IV (94 Metal - High U-236)	07	47	Initial Full Length Mk-IV Columns
G	21	Mk-IV (94 Metal - Central)	■	1	Graphite Samples Channel
N	1	Mk-1B (Natural U)			
X	244	Mk-1A (125 - 94 Metal)			
	916	Total		88	Total PTs
	88	Total PTs			
	1004	Grand Total			

*Includes Mk-IV high U-236 content fuel, 2 tubes with Mk-1A 125-94 Metal and Mk-IV 94 Metal, and 2 tubes with Mk-IV AA 125 Metal and Mk-IV 94 Metal.

Loading Pattern - N Reactor

BN-A

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FUEL AND TARGET FABRICATION - K REACTORS

PRODUCTION

General

Production of AlSi-bonded fuel for the K reactors was 104.8 percent of forecast. All of the elements had bumpers or self-supports attached. Hot-die-sizing operations continued down (see Equipment Experience, below). Thoria canning was entirely with wafered material.

Acceptable Elements Produced

	Finished Production (tons)	Yield - Percent	
		Current Month	FYTD
AlSi-Bonded Fuel	145.7	94.0	95.6
Thoria (wafered)	25.3	90.4	90.4

Month-End Inventories

	Tons
Bare Uranium Cores	1,209*
Finished Fuel: AlSi-Bonded	1,276*
Hot-Die-Sized	58
Thoria Elements	10.5

*These totals include 120 tons of cores and 130 tons of finished fuel made for the smaller reactors.

OPERATING EXPERIENCE

Canning line efficiency of the AlSi-bonding lines was 98.6 percent. Downtime was assigned 50 percent to equipment malfunctions and 50 percent to operations. AlSi canning operations continued on the basis of two lines on the day shift, five days per week.

Considerable downtime was incurred in the thoria wafer pressing and sintering operations, due primarily to difficulty with the hydrogen-atmosphere sintering furnace. However, reactor requirements will be available.

EQUIPMENT EXPERIENCE

AEC directive EQT-106 authorized \$395,000 for replacing the AlSi-bonding process with hot-die-sizing for the production of K reactor fuels. Current status of the work on this project (DCE-524) is as follows:

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- Arrangement and layout drawings for the nickel plating machine and the press line have been approved internally and forwarded to AEC-RL.
- The right-of-way for the new line has been cleared.
- Foundations for the nickel plating machine and the press are complete.
- The press and preheat furnace have been disconnected and moved from the 306 Pilot Plant to the 313 Building.
- The nickel plating machine has been dismantled in the 3716 Pilot Plant and is being prepared for relocation on the new foundation in the 313 Building. Serious corrosion to the machine support steel was revealed as the machine was dismantled for relocation. To halt this corrosion, the support steel (which will not be readily accessible after reinstallation of the machine) will be cleaned with solvent, steel brushed, treated with rust preventative, primed and covered with a chemical resistant coating.

PROCESS ASSISTANCE AND CONTROL

Nothing significant to report.

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FUEL AND TARGET FABRICATION - N REACTOR

PRODUCTION

Input Production

Total billets extruded	253
Tons extruded	55.4
Percent of forecast	102.6

Output Production

Total finished fuel assemblies	1,680
Tons output	32.3
Percent of forecast	82.8

<u>Uranium Utilization - %</u>	71.2
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Month-End Inventories - Tons

Bare uranium billets	208
Finished fuel	319

OPERATING EXPERIENCE

Output production was behind schedule because personnel from final assembly were diverted to other work of higher priority.

EQUIPMENT EXPERIENCE

Cut-Off Saw

The main drive shaft of the deSanno cut-off saw replaced last month failed on June 13. The break was at the junction of the 1.625" diameter and the 1.775" diameter which was the same location as the break in the original shaft. The outboard bearing operates on a sleeve that is pressed to the proper position on the 1.625" diameter section of the shaft. In each failure this bearing mount was found shifted from its assembled position, causing the end section of the shaft to be unsupported.

The original shaft, welded at the break location, was installed for use until a new shaft can be fabricated. The new shaft will be redesigned to provide a backing shoulder to prevent the bearing sleeve slippage. The saw was put back into production June 16, but limited to single-tube operation until a shaft of the new design can be procured.

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Uranium Oxide Burner

A continuous exhaust stack sampling system was installed in the oxide burner, and the filters, exhaust fan velocity, and general air flow pattern from the control room were tested. Inlet air cracks around the doors were sealed and provision made to maintain a controlled air-flow pattern throughout the building. The north burner in the facility was started on June 17. After 20 loads were burned, continuous stack measurements ranged from 2.9×10^{-11} microcuries/cc of air (max.) to under 1×10^{-12} microcuries/cc of air. Maximum allowable is 6×10^{-11} microcuries/cc of air. During this period, the pressure differential across the filter increased 1/2-inch of water but was still well within the allowable limits. Operators wear respirators because during the load-unload portion of the burning cycle the light oxide still presents a potential hazard in the control room.

PROCESS ASSISTANCE AND CONTROL

Butt Shear Development

A satisfactory tool design for the butt shearing of Mark IV outer coextrusions on a production basis has been completed. Testing this month revealed tooling performance to be excellent. Mandrel life is expected to be equal to or better than conventional mandrel life. The shear tool and follower block life expectancy should also be good, barring accidental damage caused by overheating or misalignment. Twenty-seven extrusions have been sheared with the new tooling. This, in addition to 38 extrusions processed earlier using tooling of a prototypical design, brings the total number of outer extrusions sheared to 65. Results have been excellent to date with no processing problems encountered.

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TECHNICAL ACTIVITIES - K REACTORS

RESEARCH AND DEVELOPMENT

Mission 1 - Basic Production

1-A. Brittle Fracture Program - K Inlet Piping

Technical Bases for K Reactor Tube Power Limits

The results of K reactor limit calculations for the case of dual riser loss with an E-Q load have been included in the recently published safety analysis report on the brittle fracture situation (see "Safety Analysis Reporting", below).

Refined examination of the effects of specific individual parameters, such as variations in speed of portions of the safety rod system, is being conducted to supplement the calculations made to date. Planning has been initiated for the development of limits to cover the potential range of conditions with normal uranium loadings as anticipated for the K reactors by the end of calendar 1969.

Fracture Mechanics Tests

Fatigue crack growth rate data were completed for longitudinal weld and base metal materials using single-edge-crack (SEC) specimens in process water environment. A plot which combines the WOL (wedge opening loading) and SEC data provides fatigue crack growth rates in the range of interest which are about a factor of two higher than those which were obtained from the WOL data alone. The behavior of the longitudinal weld and base metal samples was similar.

Fracture Mechanics Analysis

A fracture mechanics analysis utilizing the materials tests data and stress analyses and measurements was completed to determine proof test parameters. A proof pressure of 900 psi will provide protection against fracture from the slow growth of hidden defects in the K reactor pipe and riser materials under both normal and emergency stresses. To demonstrate continued protection, the proof tests would need to be repeated before 4000 startup-shutdown pressure cycles of the reactor occurred.

Details of this analysis are presented in the Feature Report which is appended as Section H.

Safety Analysis Reporting

Analyses of fracture mechanics data, and of analyses conducted during the comprehensive program started in late 1968, have been summarized in HW-74095 SUPP5, "Safety Analysis of Coolant Supply Piping - Hanford K Production Reactors," issued June 13. The conclusions presented are:

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- Adequate technology has been developed to permit establishing limits on reactor power such that a failure of a major coolant supply pipe will not result in a fission product release. Such limits can be derived for a variety of assumed conditions: various fuel loads, six- or five-pump operation, and either one or two coincidental pipe breaks.
- The piping contains joints where undesirably high stresses occur. Although most of these joints are in runs of ductile 24-inch pipe, they should be replaced so that the system can be operated in accordance with modern codes and good design practices.
- The periodic application of a proof pressure (900 psi versus an operating pressure of 495 psig pump discharge and 400 psi top-of-riser) will adequately demonstrate immunity against both (1) the growth of any existing defects to a size where pipe failure would occur, and (2) any foreseen overstresses of the piping which could result in initiating pipe failure at an existing small defect. This proof test is judged to demonstrate the same degree of quality assurance for this particular system, under its particular operating conditions, as would be provided by ductile, higher quality piping.

A meeting with the AEC's Division of Reactor Licensing to discuss the conclusions of the Safety Analysis Report was held at Bethesda, Md., on June 24-25.

1-B. Zircaloy Process Tube Hydriding

Effect of Thermal Cycling on Case Layer Concentration

A section of process tube from channel 2462-KW has been thermally cycled between cold water and boiling water for the past year. This tubing had a typical case layer about 3 mils thick, with 961 ppm total hydrogen and 97 ppm base hydrogen before cycling. After 1 year of cycling, the only detectable change was about 2 ppm increase in base metal hydrogen content; this small change is within the normal analytical error or variation in sampling.

These data indicate that temperature cycling, as occurs during startup and shutdown, is not the driving force that is moving hydride into the base metal. As no aluminum was present in this test, only thermal cycling was involved.

Examination of Grit-Blasted Tubes

Tubes 1354- and 1355-KW, removed for analyses on May 20, had been grit-blasted on September 10, 1966, and subsequently operated with anodized dummies. Although detailed analyses, both chemical and metallurgical, are not yet complete, samples from several locations show that, where the case

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layer had been removed completely, no further hydriding occurred. These findings agree with those from three tubes removed from KE Reactor in December 1968. In each tube, grit-blasting left some areas where case layer removal was not complete.

Tube Samples - C Reactor

The downstream 72 inches of ten of the "C-size" Zircaloy tubes at C Reactor were cut off for use in electrochemical dehydriding development. It was not possible to withdraw three of the ten tube sections, presumably due to corrosion product between the tubes and the gunbarrels. One of the seven sections which were successfully removed was partially stuck in the gunbarrel and was removed with difficulty. The Van Stone flange on this tube was found to be cracked and there was evidence of leakage. Hydrogen analyses on these tubes are not yet available.

1-C. Corrosion Studies

Production Test Authorization 176, for comparing the corrosion of aluminum-clad fuel elements in process water containing 0.5 and 1.0 ppm dichromate, was approved for KW Reactor. The fuels to be monitored were charged into the reactor on June 14.

1-D. Decontamination Studies

Production Test Authorization 158, permitting single-tube and complete-crossheader through-reactor decontamination tests, is circulating for approval. Materials which the decontamination solutions will contact during passage through the reactor are being obtained in preparation for laboratory corrosion tests.

1-E. Fuel Cladding Localized Corrosion Studies

Four columns of PT-159 (Universal Core Comparison Test) fuel were discharged at an average exposure of 600 MWD/T for interim evaluation of test performance. Twenty-eight fuel pieces from the downstream 16 positions were examined in the KE viewing facility: 13 K5E AlSi fuel, 6 K5AE large core HDS fuel, and 9 K5AE-A universal core HDS fuel. Visual examination revealed no marked difference in corrosion as a function of fuel model, and no corrosion more severe than 10 mils was observed. On the basis of these observations, the planned goal exposure of 1200 MWD/T has been scheduled for the remaining 20 test columns.

1-F. Physics Code Development

The WIGL3 two-group kinetics code could not be obtained as expected as it is not yet available to the Argonne code center. Conversation with the original code developer indicates, however, that the WIGL2 version should be adequate for overcoming the initializing problems experienced earlier with WIGL2.

Conversion of the isotope buildup-burnout code ALCHEMY from the IBM 7090 to the UNIVAC 1108 is being accomplished by BNW under a work order supplied by

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DUN. Some small improvements are being incorporated, such as a plotting routine, in the revised version which will be called ALKEMY.

Mission 2 - Coproduct

2-A. Lithium-Aluminum Splines

Analytical results on samples from LiAl spline irradiated 48 days at KE Reactor show that the tritium concentration was very close to that predicted by physics calculations. Maximum total gas content was 20 GVR (Gas Volume Ratio), including 6.66 T₂ GVR.

Mission 3 - Transplutonium Technology

3-A. Ten-Kilogram Plutonium Irradiation - PTA-150

Tube powers of the PuAl columns within the test block continued to run within about 20 percent of intended levels, and the 94 Metal columns at the edge of the block continued at normal levels. Six more thorium columns adjacent to the PuAl columns were replaced with 94 Metal during the outage late in June; there are sufficient spline tubes in the affected region to compensate any local reactivity in excess of the amount desired.

Mission 4 - Pu-238

4-A. Americium Irradiation

The 12-piece americium test loading (PTA-171) was charged into column 1678-KE on May 27, with supporting 94 Metal enrichment in two adjacent columns. Temperature readings subsequent to the charging indicate that the local reactivity and heat generation effects are consistent with predictions.

4-B. Neptunium Irradiation

PTA-163, authorizing irradiation of four neptunium columns in a central-zone cluster and one column in an outer fringe tube, was approved and the material was charged into KW Reactor on June 16. Early observations following startup indicated reactivity and heat generation rates in the range expected.

A paper on the production of "clean" Pu-238 (defined as that containing 0.3 ppm or less Pu-236) was presented at the June ANS meeting in Seattle. The demonstrated benefits of Hanford reactors were pointed out, especially when augmented by use of graphite as the target diluent and potentially with use of the cluster technique to further reduce higher energy (gamma,n) and (n,2n) reactions in Np-237.

Mission 5 - Other Isotopes

5-A. Cobalt Tubes

Tube operating flow and temperature characteristics have been satisfactory with the cobalt-bearing liner tube (Haynes alloy 25) which was installed

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in process tube 1355-KW on May 20. Cobalt analyses on water samples are not yet available.

The six napkin ring segments from one of the cobalt-aluminum sandwich tubes have been charged into a KE water-cooled test facility.

Mission 7 - Target Space Enhancement

7-A. C Reactor Overbore

The central overbore element from each of the two columns 2864-C and 3167-C has been sent to BNW Radiochemistry for isotopic buildup and burnout analyses. Comparisons can also be made with the chemical results of gamma-scanning information obtained earlier by BNW on the core rod from element 16 of column 3167-C.

7-B. K Reactor Overbore

Total control calculations have been completed for the K reactor overbore cases, and documentation of the study is nearing completion. As expected, total control considerations during flooding of the hot lattices would represent the most severe physics limitations on the extent of overboring allowable. The changes in reactivity upon water loss would be negative and large for all of the overbore fuels considered. Although in some cases the effects of subsequent lattice heating would cause a net reactivity gain, this gain would be easily suppressed by the existing control system.

Mission 10 - Columbia River

10-A. High Turbidity Coolant

The half-plant test at KE Reactor with high turbidity (0.3 JTU) water was resumed on June 9. The test had been temporarily discontinued during the spring runoff due to high turbidity on the control half of the reactor.

10-B. Effect of Ionic Impurities on Effluent Activity

Addition of silicate to the Water Treatment Pilot Plant clearwells corrected the problem of silicate buildup on the filter beds and improved turbidity. The treated water, made by addition of the major ionic river water impurities to deionized water, was routed to the in-reactor tubes on May 28. Samples of effluent water are being analyzed for radioactive isotope concentrations.

ENGINEERING AND TECHNOLOGY - REACTORS

Project Engineering - Reactor Facilities

The status of approved construction projects relating to K reactor facilities is summarized in Appendix A.

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ENGINEERING AND TECHNOLOGY - FUELS & TARGETS

Long Fuels Program

In view of the cost incentives for increasing fuel element length, a program has been initiated to make and test longer fuels. The fuel length established for preliminary testing is 11.230 inches with a 10.750-inch core. AEC-RL had been requested to deliver 2500 of these long Model K11E cores (1.420" OD, 0.545" ID) by the end of August so that hot-die-sizing and reactor charging schedules could be met. However, NLO indicates that core deliveries cannot be made until mid or late September because of long lead-times required for delivery of new tooling needed for the long-core model. About half of these cores will be irradiated in a K reactor late this year under a Production Test to determine their dimensional stability under reactor operating conditions. The remaining cores will be used for process development tests to optimize core geometry and the hot-die-size process in anticipation of conversion to this process by the end of FY-1970.

This initial order for 2500 long cores requires two types of heat treatment - one-half are to be hollow induction heated-oil quenched and half are to be solid induction heated-oil quenched. No major problems are anticipated for NLO to induction heat treat the long cores as their induction equipment was designed to accept cores up to 12 inches in length. Possibly the only modification required would be a new or modified coil arrangement; the need for this should be determined by mid-July following heat treat tests for long cores currently underway at NLO.

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TECHNICAL ACTIVITIES - N REACTOR

RESEARCH AND DEVELOPMENT

Mission 1 - Basic Production

1-A. Mark IV Fuel Development

Evaluation of the Mark IV demonstration test fuel performance data is about 50 percent complete. In all study cases, the swelling rate has been linear with exposure over the range to which they were irradiated (up to 3230 MWD/T). In general, the outer elements have swelled at a slightly greater rate than the inner elements.

Both inner and outer fuel elements have shown a linear decrease of length with exposure. The scatter in data is greater than desired for an accurate determination of the rate of change; however, in the case of the Alloy 503 test fuel, the average rate of length reduction has been about 10 mils per 1000 MWD/T exposure. Evaluation of the length data from the other study cases is in progress.

The eight remaining Mark IV monitor columns are goaled to be irradiated to high exposures for comparison with Mark I high exposure performance data; i.e., one each to 3500, 4000, 4500, and 5000 MWD/T, and four columns to 6000 MWD/T.

1-B. Mark IV AA Fuel Development

The two prototypic Mark IV AA (125 Metal inner and outer) spiked columns were charged into fringe locations during the mid-June outage, as authorized by PT-NR-94 SUP5. Column instrumentation will provide the thermal hydraulic operating data necessary to evaluate this fuel design. Normal goal exposures are planned for these columns. Assemblies fabricated with flat-cap, brazed closures were alternated in the two test columns with assemblies fabricated with chevron-cap, brazed closures. Profilometer measurements of the end closures will be compared for both the pre- and post-irradiation conditions in order to examine the effect of cap geometry on the dimensional stability of the closure.

1-C. Mark I-C Studies

The two monitor columns containing upset-forged Mark I-C outer elements have been examined and evaluated. The average swelling rate observed was 0.45 percent $\Delta V/V$ per 1000 MWD/T which is equivalent to that of previous Mark I swelling rate data. The maximum swelling rate was 1.00 percent $\Delta V/V$ per 1000 MWD/T, which also is typical of Mark I performance. The direct-cast billet fuel has been performing without incident. Three columns in Mark IV geometry are to be charged as soon as space is available.

1-D. Fuel Element End-Spider Development

Flow split tests to determine the optimum support stagger to compensate for end-spider pressure loss characteristics are almost complete with one test to be run with a 1-inch stagger. Tests will then have been performed for support staggers at 1, 2, 3, and 4 inches on the same fuel assembly. The optimum support will be calculated and a flow split test on fuel with end-spiders run to confirm the calculations.

One column of 16 Mark IV assemblies has been fabricated and assembled with end-spiders and is to be tested as soon as the optimum support stagger is determined. Minor modifications to the existing end-spider projection welder are expected to permit fabrication of the five columns of fuel for a reactor test scheduled in September.

The status of end-spider life tests in the flow loop, hampered recently by equipment problems, was as follows at report time:

<u>Fuel Type</u>	<u>Support System</u>	<u>No. Fuel Elements</u>	<u>Loop Hours</u>
26" Mark IV	End-spiders	7	4767
26" Mark IV	End-spiders with one or two legs removed	3	2990
26" Mark IV	End-spiders with one or two projection welds drilled out	2	0

One of the last two assemblies replaced the assembly which failed in May; the other displaced one of the eight assemblies, reducing that group to seven.

Fracture surfaces of end-spiders that failed in the flow-loop have been sent to BNW for electromicroscopy analysis of striations on the fatigue surface.

1-E. Unbonded End Closure Development

The successful development of a brazeless or unbonded end closure of the chevron geometry for N fuels is prompted by a substantial estimated saving in fabrication costs, combined with a potential quality improvement from corrosion considerations. This development program is an extension of the efforts to promote a flat cap unbonded closure, but differs in that a mechanical restraint between the uranium core and the chevron cap is provided.

Sixty outer fuel components have been fabricated from peel-test reject Mark IV extrusions. The test elements are about 6 inches long and have end closures of four basic geometries. End closure fabrication of 30 additional 6-inch fuel elements from reactor-grade extrusions has been initiated.

Ex-reactor testing plans for the test closures include standard beta-heat treat cycles (room temperature $\rightarrow$ 732 C $\rightarrow$ 300 C $\rightarrow$ room temperature), autoclave (100 F $\rightarrow$ 750 F $\rightarrow$ 100 F) under 1500 psi hydrostatic pressure, load deformation measurement, and impact tests. The test results with the unbonded end

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closures are to be evaluated with respect to the behavior of flat cap brazed closures subjected to identical treatments. The reactor testing program phase will be initiated pending the complete ex-reactor characterization of both inner and outer unbonded closure properties.

1-F. Zircaloy Process Tube Surveillance

Hydrogen analyses of process tube sections covering the length of process tube 0758, discharged in February, have been determined. The hydrogen contents at the upstream spacer region were lower than those found for tube 2455, discharged in November 1967 (50 ppm versus 70 ppm). The lower total hydrogen content is even more significant considering that the original hydrogen content of tube 0758 was 28 to 42 ppm whereas that for tube 2455 was 8 to 9 ppm. These results indicate that hydrogen buildup has not been occurring with continuing operation, reversing the inference possible from comparison of data from tube 2455 with that from tube 1756, discharged in May 1966.

Preliminary mechanical strength data from 0758 tube sections indicate equivalent strength to tube 2455 examined previously. The exposure of 0758 was calculated to be 61 percent higher than that of 2455.

1-G. Graphite Studies

Design proceeded on a wedge tool to allow unloading of the remaining graphite blocks from ball channel 60 to be completed during the scheduled July outage. This tool is to be used to remove the projecting obstruction which prevented removal of the graphite samples on schedule. A review of the observations and graphite layup drawings has led to the conclusion that only one graphite block is projecting into the channel. Previously it had been conjectured that the stack may have shifted at the top process tube layer in a front-to-rear direction.

1-H. Corrosion Studies

Control Rod Pitting

The two seal section aluminum tubes removed from N Reactor control rod 14 during the May outage after chromic acid cleaning were sectioned and examined for pitting. Twelve scattered pits 40 to 80 mils in diameter were found on the two tubes (combined length about 64 feet). Shallow pits up to 10 mils deep were free of corrosion products; the deeper pits were partially filled with corrosion product, indicating the pits existed prior to, and were not the result of, the chromic acid cleaning.

Five of the pits are now being examined metallographically to determine pit depth and geometry. An additional control rod seal section will be removed without cleaning during the July outage for further inspection.

Specifying of GCR Decontamination Conditions

DUN-5876, "Operating Conditions and Requirements for Decontaminating an N Reactor Graphite Cooling System Heat Exchanger at Building 189-F", was

issued to specify the decontamination conditions and limits for decontaminating the No. 4 GCR heat exchanger. Decontaminating and retubing are tentatively scheduled to begin early in July.

1-I. Process Surveillance Systems Study

The engineering study for a process surveillance system has been largely completed. Work has been finished on the installation and maintenance requirements of the computer system itself, information has been compiled but is yet to be documented on data acquisition requirements of the applicable plant buildings and major systems, and the study of reliability-availability, automatic checkout, and prime power aspects is well along. A base case model for the "surveillance system fault tree" was constructed during the report period.

1-J. Simulation of Primary Loop Effects

A MIMIC code simulation of the primary coolant loop control systems has been made to determine the response to a "worst case" scram transient as a function of the number of HPIPs (High Pressure Injection Pumps) available. A simulation was also made to determine the response of the system to a scram with the HPIP speed control in the manual mode (fixed fluid coupler scoop tube position). The results of these simulations are still being analyzed.

1-K. ISI-609 Computer Programming

Debugging of the input-output control and calculational routines continued during the report period. Twenty-four of the 25 programs developed are now operational. The remaining debug efforts have been held up due to hardware malfunctions in the 609 logic.

Mission 5 - Other Isotopes

5-A. Cobalt Irradiation

The cobalt elements which became lodged in graphite cross-coolant channel 1588 during the May outage continued to be lodged in that tube. Physics calculations indicated that cobalt release from the sample would be small even if the cladding were breached; furthermore, reported results of the chemical analysis of crud samples from tube 1588 indicated that the target cladding had not been breached. Plans are being made to discharge the cobalt from this channel in July.

Mission 7 - Target Space Enhancement

Graphite Coating Studies

Coating studies with TMS (tetramethylsilane) and TES (tetraethylsilane) were continued to characterize coating parameters. Oxidation tests with recoated samples from these experiments, using standardized oxidizing conditions (helium containing 6000 ppm H₂O at 900 C) for periods ranging

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between 70 and 150 hours, have shown that oxidation rates have been reduced by the coatings by factors ranging from 3 to 15.

Zr-Graphite Compatibility Loop Studies

Zircaloy-2 samples exposed simultaneously in the loop while coating graphite samples with TES were found to have a silaceous coating on their surfaces. Hydrogen analyses, as determined by optical means, did not indicate hydriding. The study is being continued.

Mission 8 - Nuclear Safety

8-A. Metal-Water Reaction and Fission Product Release Studies

Operation of the 32⁴ Building "D" cell was inaugurated by BNW on June 5. Two tests have been conducted utilizing a high-exposure (3000 MWD/T), short-cooled element to permit measurements of fission product release. Analysis of the data obtained from both tests on metal-water reaction rates was initiated. Laboratory analyses of the fission product releases also are underway.

In the initial test, internal fission gas pressure resulted in substantial swelling of the fuel element at temperature although no uranium extruded from the clad. In the second test, an external quartz tube simulated a process tube; extruded uranium completely plugged the annulus.

ENGINEERING & TECHNOLOGY - REACTOR

V-12 Diversion Valve Study

The V-12 valve study continued to receive maximum attention, with major emphasis on the selection of corrosion resistant materials from which to fabricate the actuator. A procurement request for actuator body castings of Inconel has been processed. Various other aspects of the assembly are also being or have been investigated for compatibility with the environmental conditions in which the V-12 valve must now operate.

Ball Trip Mechanism

Two prototype ball hopper trip mechanisms of a new design were fabricated and assembled in the 1705-N Development Lab for testing and evaluation. After preliminary trials, a decision was made to concentrate on the direct-coupled design and abandon the lever arm concept. Finalized design and selection of corrosion resistant materials for the clutch and the armature plate are in progress.

Project Engineering - Reactor Facilities

The month-end status of approved construction projects relating to N Reactor facilities is summarized in Appendix A.

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DN-561

ENGINEERING & TECHNOLOGY - FUELS & TARGETS

Outer Support Development

An evaluation was conducted to determine the effect of material grain size on N fuel outer support performance characteristics. The objective was to show that the purchase specification calling for a maximum ASTM grain size of nine was unnecessary.

Supports manufactured from material lots having grain sizes ranging from ASTM 7.5 to 10 were tested for strength and deflection characteristics. All supports tested exhibited fully acceptable strength and deflection properties, and it was concluded that ASTM grain size within the range tested is not a significant factor. A proposal to change the ASTM grain size in the purchase specification from nine to eight was prepared and approved. This change allows the offsite purchase of material without the added costs incurred by an unnecessarily restrictive specification.

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IRRADIATION SERVICES

FUEL TECHNOLOGY

(U-Pu)C Unrestrained Swelling Experiment - United Nuclear

One of the (U-Pu)C swelling experiments was discharged from a KW front-to-rear test facility. This experiment was removed prior to goal exposure due to fuel container failures in two of the four subcapsules.

UO₂-Mo Fuels Development - NASA-Lewis Research Center

Capsule 37 was discharged from the 4B general purpose facility at KW after attaining goal exposure.

Triple capsule 4, 5, and 6 was discharged after capsule 5 had experienced a fuel cladding failure in the 2C general purpose facility at KE.

Capsule 32 was discharged from the 2A general purpose facility at KE after a cladding failure occurred.

Fast Reactor Oxide and Nitride Fuels - Battelle-Northwest

The third in a series of (U-Pu)O₂ fuel development capsules was irradiated in the KW Snout facility. The fuel pins in this capsule reached the desired center-melt temperature.

MATERIALS DEVELOPMENT

Irradiation Damage to Reactor Metals - Battelle-Northwest

A boat containing nickel-base Charpy impact specimens was discharged from C Reactor's E test hole.

One creep capsule was discharged and another charged into the rod room side general purpose facilities at KW Reactor.

ISOTOPE PRODUCTION

C-14 Production - Oak Ridge National Laboratory

Thirty-four Be₃N₂ pieces were charged into a front-to-rear general purpose facility at KE Reactor.

ROUTINE IRRADIATIONS

Forty-three Quickie activation analysis capsules were irradiated in the K reactors for BNW.

IRRADIATION STATUS SUMMARY

The quarterly report on the status of all in-reactor service irradiations is presented on the next two pages.

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STATUS OF SPECIAL IRRADIATIONS
(As of 6/24/69)

<u>Test No.</u>	<u>Material or Capsule Content</u>	<u>Experimenter</u>	<u>No. of Samples or Capsules</u>	<u>Date Charged</u>	<u>Test Facility</u>	<u>Accumulated Exposure-nvt</u>
006	Graphite Burnout Samples	DUN	20 20	10/15/68 12/8/68	1880-KW 3066-KE	2.6 x 10 ²¹ 1.8 x 10 ²¹
041	Haynes Metal - Tensile Specimens	DUN	1 6	1/8/68 5/28/69	2B-KE 3865-KE	5.0 x 10 ²¹ 2.6 x 10 ²⁰
053	Graphite Coating Studies	DUN	1	Short Term	2B-KW	Various
172	Various Activation Analyses	BNW	89	Short Term	2D-KE 2A-KW	Various
184	Washington-Designated Program	BNW	107	Short Term	2D-KE 2A-KW	Various
177	Graphite Samples	BNW	8 3 9 1	9/13/68 9/20/67 1/25/69 3/14/69	B-C 3A-KE 4B-KE 2B-KW	2.8 x 10 ²¹ 4.4 x 10 ²¹ 1.1 x 10 ²¹ 1.1 x 10 ¹⁹
236	Structural Materials - Creep Rate Tests	BNW	1 1 1	6/11/68 1/14/68 1/14/68	3C-KW 3C-KW 3D-KW	2.9 x 10 ²¹ 4.4 x 10 ²¹ 3.3 x 10 ²¹
243	Irradiation of Various Reactor Structural Materials	BNW	2 1	10/24/68 2/13/69	4C-KW 4C-KW	1.6 x 10 ²¹ 9.9 x 10 ²⁰
271	Various Calibration Standards	BNW	8	Short Term	2D-KE 2A-KW	Various
381	SF Foil Irradiation	BNW	4	4/21/69	0065-KE	8.7 x 10 ¹⁸

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Test No.	Material or Capsule Content	Experimenter	No. of Samples or Capsules	Date Charged	Test Facility	Accumulated Exposure-nvt
384	Boron Carbide Irradiation	BNW	3	4/21/69	3C-KE	5.3×10^{20}
			3	5/27/69	3B-KW	1.3×10^{20}
391	LMFBR Oxide Fuel Irradiations	BNW	3	Short Term	2B-KW	Various
395	Neutron Dosimetry Studies	BNW	1	Short Term	2A-KW	Various
396	Structural Materials - Creep Rate Tests	BNW	1	5/27/69	3B-KW	1.3×10^{20}
510	Uranium-Zirconium Hydride NAA-121-1 SNAP-8 Prototype Fuel	NAR	4	10/17/68	0074-KE	5.3×10^{20}
E-4 603	Potassium Chloride - Cl-36, S-35, & K-40 Production	ORNL	80	1956 to 1967	2D-KE	3.3×10^{22} to 4.9×10^{21}
604	Beryllium Nitride - C-14 Production	ORNL	40	10/15/68	1949-KW	2.6×10^{21}
			40	7/11/67	1791-KW	7.6×10^{21}
			40	11/26/67	2160-KE	6.4×10^{21}
			40	12/2/67	2864-KE	6.3×10^{21}
			40	11/26/67	4268-KE	6.1×10^{21}
			34	5/28/69	3865-KE	2.6×10^{20}
605	Thallium Oxide - Tl-204 Production	ORNL	12	11/3/63	2D-KE	1.4×10^{22}
			12	10/22/66	2D-KE	7.0×10^{21}
700	Molybdenum-Uranium Cermet Fuel Specimens	NASA-LRC	1	6/13/68	2D-KW	2.2×10^{21}
807	U-PuC Irradiation	UNC	5	2/8/69	0074-KW	9.6×10^{19}

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ADMINISTRATION - GENERALFIVE-YEAR SUMMARY FORECAST - 02 PROGRAM FY 1971-1975

In response to a request from AEC-RL, a document was prepared outlining Douglas United Nuclear's capital and operating funding requirements for FY 1971 through FY 1975. The capital portion of this forecast was based upon the previously issued Five-Year Plan as modified to reflect current plans for N Reactor reliability improvements. The operating portion was based upon DUN's FY 1971 operating budget, as revised to reflect changes in production assumptions, additional operating funds required for N Reactor reliability improvements, and continuation of the noble gas release studies.

NAVAL HEAT EXCHANGER PROPOSAL

A Request for Bid was received from Westinghouse Electric, Inc., Apparatus Division, to cover a proposal for the decontamination and inspection of one naval heat exchanger and the storage of an additional unit. Due to the potential work in this field, an agreement was reached between AEC-RL and DUN to submit a proposal under DUN's Prime Contract. Officially, this arrangement has the effect of the Atomic Energy Commission submitting the proposal to the Department of the Navy. The preliminary cost estimate submitted for this work was \$967,000.

DISPOSAL OF SURPLUS FACILITIES - F & H AREAS

Pursuant to the AEC disposal of surplus Government property in the 100-F and 100-H reactor plants, dismantlement of the equipment in 190-F Annex is continuing and dismantlement of the 184-F and 184-H steam generation buildings and equipment is just beginning. Interested buyers for equipment in 181-F and 182-F are inspecting the facilities; bid opening date is July 28.

APPROVAL LETTERS

At the close of the reporting period, final AEC-RL action was pending for the following requests:

<u>ATD Number</u>	<u>Subject</u>	<u>Date of Transmittal to AEC-RL</u>
AP-39	Pension Plan portion of letter entitled "Pension Plan, Salaried Savings Plan, and Wage Savings Plan"	January 12, 1966
ATD-156 Add. #3 Sup. #1	Education and Training Programs Educational Leave of Absence Program	June 24, 1969
ATD-180	Lay Off for Lack of Work Appendix "B" Modification MO-23	February 24, 1969

MANAGEMENT DATA PLOTTING SYSTEM

DUN agreed to participate with ARHCO and ITT/FSS in authorizing the development of a generalized plotting system by Computer Sciences. The system will provide a software interface between the Univac 1108 and the Calcomp plotter, and will eliminate the need for a special program each time an organization requires a new type chart to be drawn by the Calcomp plotter. Total costs for programming are estimated at \$8,125, of which one-third will be charged to DUN. The system is planned to be operational by September 1, and will be available for use by all components for chart preparation work.

BUDGET CALCULATOR SYSTEM

A budget calculator model was developed and programmed for use as an aid in budget preparation. Various conditions may be fed into the model to reflect changing modes of operation with resultant costs generated by the program. The budget calculator program is operational following detailed analysis of test runs which used a representative selection of actual data.

Further refinements of these programs may be required as more extensive data are used and the user develops more sophisticated requirements. Based on execution times noted during the tests, the computer time required for a complete processing should be less than five minutes.

PLANS FOR PROGRESS - EEO

Two Negro Co-op students were employed on June 9. Both have been accepted at Washington State University and will enter school this fall. The Company now has four minority employees who are participating in the Cooperative Education Program.

FORCE REDUCTION STATUS

During the period January 1 to June 30, the Company's work force was reduced by a total of 164 employees. Forty-seven of the 164 were by attrition prior to the announced shutdown of C Reactor. An additional 57 left by voluntary layoff, leaving only 44 who were laid off involuntarily. Twenty-four of these were placed with other Hanford contractors.

SAFETY

No personnel radiation exposures exceeded operational control.

Month-end safety statistics were:

Disabling injuries - June	0
- Year to date	0

Days since last disabling injury 242

Man-hours since last disabling injury 2,418,000

EMPLOYMENT SUMMARY

DUN personnel totals and employee allocations as of May 31 and June 30 are shown in Appendix B.

APPENDIX A

PROJECT STATUS SUMMARY - REACTOR FACILITIES

<u>Number & Title</u>	<u>Authorized Funds - \$</u>	<u>Percent Complete</u>		<u>Status</u>
		<u>Design</u>	<u>Construction</u>	
<u>K Reactors</u>				
DCE-505, Boiler Control Improvements - 165-KE & KW	410,000	100	80	Acceptance testing of No. 2 KE boiler was completed June 19 and beneficial use attained. Equipment installation continued on the No. 2 KW facility which is about two-thirds complete.
DAP-510, Discharge Chute Clearing Equipment - K Reactors	220,000	100	96	AEC Directive No. AEC-294, Mod. No. 1 increased the authorized funds by \$20,000 to a new total of \$220,000, with a completion date of November 30. Modification of the KE bucket guides is in progress.
DAP-516, Storage Building Addition - 105-KE & KW	142,000	5	0	Design work resumed on June 13, when a release to proceed with the project was received from AEC-RL.
DAE-518, Effluent Radio- iodine Monitor - KE & KW Reactors	100,000	100	0	Bids for the electronic modules for instrument assembly have been received, reviewed, and purchase recommendations made.
DCP-522, Modification of Reactor Coolant Crosstie Piping - 105-KE & KW	163,000	100	60	Installation at KW was completed May 29. Installation at KE is scheduled to start July 7.

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PROJECT STATUS SUMMARY - REACTOR FACILITIES (contd.)

Number & Title	Authorized Funds - \$	Percent Complete		Status
		Design	Construction	
<u>N Reactor</u>				
GCP-406, Improved Safety Platforms and Accesses - 100-N Area	300,000	100	96	Installation of reactor platforms being accomplished as outage time permits. Project proposal revised to redefine scope of work and extend completion date; internal review is in progress.
GCE-408, W, C & D Elevator Safety Improvements - 105-N	90,000	100	33	Revised project proposal extending the completion date to November 30 submitted for AEC-RL approval. Installation scheduled for July.
GCP-411, Effluent Control Program - 100-N Area	1,830,000	100	22	Work progressing on Sections I, II, & III. Silo poured to elevation 491' on June 20. Fabrication of charpy-tested pipe for 182 header completed June 19. Shop work on header fabrication in progress. Revisions to design are in progress for Section IV, Diesel Fire Protection. Contractor detail design drawings and equipment specifications are in the process of being approved for the Emergency Dump Tank. Foundation design problems were resolved in meeting with American Pipe Co. on June 26.
DCE-519, Replacement of Bridge Crane and Hoist System with New Crane System - 105-N Storage Basin	269,000	40	0	Project proposal revised to reflect the new \$400,000 estimate caused by preliminary crane quotes and to extend completion date; internal review is in progress.

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APPENDIX BEMPLOYMENT SUMMARY
(with employee allocations)

	<u>5/31/69</u>	<u>6/30/69</u>
<u>CONTRACT PERSONNEL</u>		
<u>02 Programs</u>		
Douglas United Nuclear	1735	1698
Assisting other Contractors	<u>14</u>	<u>14</u>
Total - 02	1749	1712
<u>Other Programs under AEC Contract</u>		
Assisting other Contractors and WPPSS	43	43
Special Irradiations	<u>6</u>	<u>6</u>
Total - Other Programs	49	49
Total Contract Personnel	<u>1798</u>	<u>1761</u>
<u>COMMERCIAL ACTIVITIES PERSONNEL</u>	23	15
TOTAL FORCE	<u>1821</u>	<u>1776</u>

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FEATURE REPORT

THE APPLICATION OF FRACTURE MECHANICS TO AN ASSESSMENT OF K REACTOR PIPING INTEGRITY

INTRODUCTION

Current nuclear safety criteria for the acceptance of reactor coolant system components require that they be made from materials with nil ductility transition (NDT) temperatures at least 60 F below the minimum operating temperature to assure that if component failures occur in service they will be ductile rather than brittle in character. Brittle failures can result in complete and sudden loss of a component; if this occurred in an inlet riser or related piping of the K reactors, reactor coolant supply could be seriously impaired.

When tests on samples of the K piping indicated a transition zone in the normal operating temperature range, several precautions were taken at both KE and KW (effective 9/1/68) to reduce the brittle fracture potential until its seriousness could be assessed. As principal steps, the reactor riser pressure was lowered 10 percent and new tube power limitations were adopted which would prevent fission product release in the event of a riser failure. A complete summary of the situation at that time, with planned evaluative studies and possible corrective measures, is contained in DUN-4709 issued 9/19/68.

One of the studies initiated was the determination of whether criteria other than the NDT plus 60 F could provide the necessary assurance that brittle failures would not occur during reactor operation. It was decided, with the aid of consultants, that fracture mechanics test data and theory applied in conjunction with proof testing could provide this assurance. The purpose of this report is to summarize the results of the fracture mechanics study, as applied to the assessment of K reactor inlet piping integrity.

THEORETICAL CONSIDERATIONS

The initiation of a fast fracture is controlled by a complex interrelationship between defect size and stress state in the vicinity of a defect. A necessary condition for fracture to occur at nominal stress levels below the yield strength of the material is that a defect of sufficient size be present; without a defect or stress concentration present, failure cannot occur at nominal stresses below gross yield.

The critical flaw concept states that for each flaw size there is a critical stress which will cause an unstable fracture to initiate. Two mechanisms can act on a subcritical flaw in the K reactor piping to make it become critical; namely, (1) slow growth to the critical size at normal operating stress levels, promoted by some external influence such as corrosion or fatigue, or (2) the sudden application of an emergency stress level which exceeds the critical stress for fracture of existing defects. An assessment of the K reactor piping, therefore, requires a determination of: (1) the size of defects

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existing in the system; (2) whether these defects can grow in service and, if so, by what mechanisms and at what rates; and (3) how long it would take for existing defects to grow to the critical size under the action of relevant growth mechanisms.

The theoretical treatment of the relationship between a defect's size and its failure stress belongs to the discipline called fracture mechanics. The bases of a fracture mechanics analysis relate the significant fracture parameters (defect size and stress) to a material property designated the critical stress intensity factor, K_{Ic} . Having identified this property, the increase in defect size and/or elevation in stress necessary to initiate fracture can be calculated. K_{Ic} is a basic material property under plane strain conditions and is related to the energy for unstable crack extension. With less restraint, such as in a sufficiently thin component (the K reactor piping is an example), the effective toughness of the material may be greater than K_{Ic} . The thickness-dependent toughness of a through-cracked plate is referred to us as K_c . Conservative values of K_c were obtained for this study by making measurements at temperatures below the operating range.

In addition to fracture analysis, fracture mechanics is well suited for evaluation of fatigue crack growth rates. It is generally accepted that crack growth by cyclic fatigue is related to the plastic strain range. Since the stress intensity factor is a measure of plastic strain at the crack tip, good experimental correlation is found between growth rate and stress intensity factor, at least in the range where linear elastic fracture mechanics is applicable.

Crack growth rate as a function of stress intensity factor is determined experimentally by cycling a pre-cracked specimen and periodically recording the crack length. Growth rate is usually expressed as dc/dN ; i.e., change in length per cycle (inches per cycle). The range of stress intensity, ΔK , is determined from the range of applied stress and the overall crack length using the appropriate equation for stress intensity factor for the specific fatigue specimen being tested.

For some materials, defects will grow when subjected to static loads if aggressive environmental conditions are present. There is particular concern in the application of high strength materials, for example, which have been shown to be susceptible to environmental effects such as hydrogen embrittlement, stress-corrosion cracking, strain aging, and corrosion fatigue. These effects decrease in magnitude with decreasing steel strength and environment aggressiveness. They should be insignificant for the K reactor risers and pipe materials in the process water environment. Examination of the internal surfaces of the inlet piping shows that the corrosion over the 14-year operating period has been so low that there has been no measurable loss in wall thickness. Moreover, significant stress corrosion cracking is not known to occur in low-strength carbon steels in mild water environments, and strain aging embrittlement due to carbide precipitation does not occur below 300 F (maximum K reactor inlet coolant temperature is 70 F).

Based on the foregoing, it was concluded that cyclic fatigue would be considered to be the primary mechanism for crack growth in the K reactor piping.

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FRACTURE MECHANICS TESTING PROGRAM

Sampling Procedure

The upper sections were cut from the D and A risers at KE Reactor and the B riser at KW Reactor for fracture mechanics tests. A typical riser section, marked with the locations from which laboratory specimens were machined, is shown in appended Figure 1. Test specimens were cut from the base metal, the longitudinal and circumferential weld metals, and heat affected zones.

Critical Stress Intensity Factor

Wedge-opening-loading (WOL) specimens were machined from the riser sections and then were tested (by the Lawrence Radiation Laboratory and Battelle-Northwest) in accordance with the recommended procedure ASTM-STP-410. The cyclic loading rate used for introducing the initial sharp crack was somewhat higher than the recommended value for some of the specimens. Test results confirmed that this difference did not affect the K values obtained. In addition, the 1-1/4 inch wall thickness of the K reactor piping was not sufficient to meet the ASTM requirement for K_{Ic} testing. Conservative K_c values were obtained for this study, therefore, by testing at temperatures 110 F below the operating range.

After machining, each WOL fracture sample was fatigued to introduce a sharp crack at the root of the machined notch. Cracks were grown to 0.10-0.15 inch in length in a zero-tension-zero cyclic loading schedule. After fatiguing, the samples were pulled in tension. The critical stress intensity factor to initiate fracture was calculated using an appropriate equation developed for the specimen geometry from the load at failure and the crack length.

The K_c values calculated from the results of the tests on the WOL specimens are plotted in Figure 2. Results are shown for the riser base metal, the heat affected zones, and the longitudinal and circumferential weld metals.

At -76 F, the specimens fractured with no apparent yielding in the plane of the fracture, indicating that plane strain conditions were probably satisfied for the sample thickness. The -76 F data are considered valid K_{Ic} values for the operating range for purposes of this application.

The D riser data points form a lower-bound curve for all the test data. The data points for the three risers sampled varied 17 percent at -76 F, 13 percent at 33 F, and less than 5 percent at room temperature. Based on these results, the minimal critical stress intensity factors for the K piping in the operating range were established as 45,000 psi-in^{1/2} for the base metal and 50,000 psi-in^{1/2} for the weld metal.

Fatigue Crack Growth Rates

WOL and single-edge-crack specimens were cracked by fatigue to sharpen the tip of the starter notch and then were cycled at constant load until the crack progressed sufficiently across the sample to obtain an adequate range of experimental data. At various points during the crack growth process, the

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Figure 2. Fracture Toughness (K_C) vs. Temperature - K Reactor Inlet Piping

stress intensity factor range (ΔK) was calculated to obtain data points relating the measured crack growth rate to ΔK . These data are plotted in Figure 3 for parent, weld, and heat-affected zone (HAZ) materials in air and water environments. The behavior of the materials was, for practical purposes, identical. The test results are in good agreement with published data on other mild steels.

STRESS CONSIDERATIONS

Normal Conditions

A summary of the operating stresses in the K reactor pipe and risers under normal loading (as used in this study) is shown in Table 1, appended. The highest stress for initiating fracture occurs in the longitudinal welds because of residual stresses which are conservatively predicted at 4000 psi in the circumferential direction. Use of the highest stress in the long runs of piping provides for lower-bound values of safe operating periods because crack growth rates are higher at higher applied stresses. Although the combined stresses at the nozzle-to-header joints are slightly higher (No. 6 in Table 1), the width of the high-stress zone in this case is small compared to the critical crack length for fracture, and the effective stress for fracture would be the lower riser wall stress.

The K reactor piping is also subjected to a 180 cycle-second cyclic stress of 100-300 psi amplitude (from pump effects) superimposed upon the normal operating stress. This class of cyclic stress results in negligible calculated crack growth (Ref: ORNL-NSIC-21). The primary mechanism for crack growth is considered to be fatigue induced by pressure cycles during reactor startup and shutdown. For conservatism, the combined stresses, including the residuals, are assumed in computing the stress intensity factor range for evaluating the crack growth rate.

Pressure Transients

Pressure transients set up during normal operation of the plant are small and limited to the normal system pressure. No quick-acting, mechanically actuated valves are present in the system. Flow transients through check valves normally are very slow in relation to the check valves closing time.

The pressure transient of significant concern is the transient following a major pipe break accident. Relatively rapid flow transients occur, including a flow reversal through a 24-inch check valve. The analysis of the pressure wave which occurs during this event indicates that this is the maximum emergency stress which could occur in the piping system. Only the initial shock wave is of concern, because of the damping of the system by the crossheader connections. The pressure transient analysis shows a peak internal pressure increase of 360 psi in the 36-inch supply piping and riser. This reflected pressure pulse is added to the transient pressure calculated for the time of valve closure which at 310 milliseconds is 230 psi at the top-of-riser, or about 250 psi at the other end of the 36-inch pipe. Hence, the maximum internal pressure in the 36-inch pipe during the transient is 610 psig.

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Figure 3. Fatigue Crack Growth Rate Data for
K Reactor Riser Materials

PROOF TESTINGProof Test Design Considerations

If the combination of defect size and stress were known throughout the system, it would be possible to calculate individually the increase in defect size and/or elevation in stress necessary to initiate fracture utilizing the K_{IC} data reported in the previous section. Such an assessment would, however, necessitate extensive nondestructive testing to locate and characterize all defects. This is considered impractical because of the physical size of the system and the limitations of testing methods. Instead, the approach being taken is to perform a "proof test" to demonstrate that any defects present in the piping system are sufficiently below the critical size required to initiate failure, at both normal and emergency stress conditions, that they can be allowed to grow between periodic test periods without danger of precipitating fracture. The proof test must be designed to protect against the following initial crack configurations and growth mechanisms:

- A through-the-thickness crack of less than critical length, which grows to the critical length for fracture.
- A partial thickness defect of less than the critical through-thickness length, which grows in service to become a through-thickness crack and then grows longer until it reaches the critical length.
- A partial thickness defect of length exceeding the critical length of a through-thickness crack, which grows preferentially in the thickness direction and causes fracture when it becomes of sufficient depth to break through the remaining ligament, either by tensile instability or by exceeding the critical stress intensity factor at the crack tip.

Through-Thickness Cracks

The stress intensity factor (K) for a through-the-thickness crack in a cylinder may be calculated using equations reported in NASA-CR-73709 (October 1968). Using these relationships, the stress intensity factor ranges for various through-thickness half crack (c) lengths were calculated for cyclic loadings of 6,000 and 12,000 psi. These data are plotted in Figure 4.

Using the minimum critical stress intensity factor for the K reactor piping of 45,000 psi-in^{1/2}, the critical crack length at 6,000 psi would be 11.7 inches. The number of cycles required to extend the through-thickness crack from any given value to the critical length of 11.7 inches at 6000 psi, may be estimated using data plotted in Figures 3 and 4 as shown in the following table:

$2c$	$\Delta K, \text{psi-}\sqrt{\text{in}}$	Average ΔK	Δc	$\frac{dc}{dN}$ in/cycle	$\frac{dN}{dc}$	$\frac{dN}{dc}(\Delta c)=N$	Accumulated cycles, EN
11.7	45,000	44,000	0.20	50×10^{-6}	2×10^4	4,000	4,000
11.3	43,000

Figure 4. Stress Intensity Factor for Through Thickness Cracks in Pipe

<u>2c</u>	<u>$\Delta K, \text{psi-}\sqrt{\text{in}}$</u>	<u>Average ΔK</u>	<u>Δc</u>	<u>dc/dN in/cycle</u>	<u>dN/dc</u>	<u>$\frac{dN}{dc}(\Delta c)=N$</u>	<u>Accumulated cycles, ΣN</u>
		42,000	0.20	40×10^{-6}	2.5×10^4	5,000	9,000
10.9	41,000	40,000	0.20	32×10^{-6}	3.12×10^4	6,240	15,240
10.5	39,000	38,000	0.25	24×10^{-6}	4.16×10^4	10,400	25,240
10.0	37,000						

It will be seen that about 25,000 cycles would be required to extend a 10-inch long crack to the critical length of 11.7 inches at the 6000 psi applied stress.

Since the stress intensity factor is proportional to stress,

$$\frac{K_{\text{critical}}}{K_{\text{initial}}} = \frac{\text{Critical stress}}{\text{Initial stress}} = \frac{\text{Critical pressure}}{\text{Initial pressure}}$$

the initial pressure can be considered equal to the operating pressure.

The ratio $K_{\text{critical}}/K_{\text{initial}}$ may be plotted as a function of cycles to failure to provide a basis for choosing a proof pressure to protect against failure from the growth of through-thickness cracks. Figure 5 is such a curve for an applied stress of 13,000 psi.

Since original startup, the K reactor piping has experienced approximately 1500 pressure cycles, or approximately 100 cycles per year, because of reactor startups and shutdowns. The current rate is 35-45 cycles/year, reflecting improved fuel performance. Considering the small number of cycles per year and the low rate of crack growth per cycle, a relatively minor overpressurization adequately demonstrates protection against this source of fracture initiation.

Short Partial-Thickness Defects

The growth of partial-thickness defects to a size critical for fracture initiation is more complex than a through-thickness crack because of many different possible crack geometries and two types of failure mechanisms. Partial-thickness cracks grow along their border at a rate proportional to the stress intensity factor. For example, Figure 6 illustrates the typical growth pattern of a small surface flaw through the wall thickness.

Experiments show that small flaws grow in a semicircular pattern. Thus, as shown in Figure 6, the minimum crack length that will just penetrate the wall thickness might be expected to be twice the thickness (2T) in length. Considering the minimum K_c values measured for the K piping (Figure 2) and the maximum stress levels shown in Table 1, the critical through-thickness crack size is seven inches (5.6 T). Therefore, small partial thickness cracks would become detectable leaks before reaching the critical size for fracture.

Figure 5. Proof Pressure to Demonstrate Through-Thickness Crack Growth

Figure 6. Typical Growth Pattern of a Small Surface Flaw Through Wall Thickness

Long Partial-Thickness Cracks

In a welded structure it is possible to have defects that are long compared to the critical length of a through-thickness crack. The growth of these defects will be in the thickness direction and at a rate proportional to the stress intensity factor at each point along the length of the defect. Since this type of defect is associated with lack of fusion or lack of penetration during welding, the characteristics of the defect could vary markedly along its length resulting in irregular localized growth; therefore, these defects may not advance uniformly through the wall to cause fracture. Rather, localized sections of a long crack would preferentially advance through areas of poor bonding, inclusions, etc., to become an observable leak. However, the concept of a long deep crack of uniform depth cannot be discounted and the growth of such a crack through the thickness could lead to fracture of the K piping.

To develop proof test conditions for the long partial-thickness flaw, the minimum size defect to be protected against must be determined. As discussed above, defects of less than three inches in length represent no hazard with respect to fast fracture. Longer cracks with depths less than 60 percent of the thickness do not represent a hazard because of the extremely long periods required for them to grow to critical size (calculated growth rates are $<10^{-6}$ inches/cycle). The defects representing a potential hazard are those with lengths exceeding the critical length of a through-thickness crack, and deep enough so that continued growth could conceivably lead to fracture. At the minimum K_{IC} value for the piping of 45,000 psi-in^{1/2}, through-the-thickness critical crack length at the combined stress in the 36-inch piping welds is seven inches at normal operating pressures.

The critical depth of a crack seven inches long based on the critical stress intensity factor is calculated by the appropriate relationship to be ~ 1.15 inches (the remaining ligament below the crack is 0.1 inch). Tensile instability would occur when the stress in the ligament reached 45,000 psi for the K reactor riser and pipe materials. This occurs when the crack depth is 1.05 inches (the remaining ligament below the crack is ~ 0.2 inch) and thus represents the limiting case.

Selection of this tensile instability mode of failure for this piping analysis is based on experiments performed by the General Electric Company to determine the failure behavior of pipes containing partial-thickness flaws of uniform depth and various lengths (Ref: GEAP-5680, June 1968). These data show that the failure loads for cracks exceeding six inches in length coincided with the onset of tensile instability in the ligament at the base of the flaw. Failure loads for smaller defects at a given depth were higher.

The proof test must provide assurance, therefore, that no long partial-thickness cracks exist in the K reactor piping that could be expected to grow to a critical depth of 1.05 inches. However, to permit a meaningful assessment of growth rate, the stress in the remaining ligament of existing flaws should be below the biaxial yield strength and the crack tip under elastic constraint when the pipe is subjected to normal operating pressure.

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Upon the advice of consultant W. W. Gerberich, a limiting ligament thickness of 25 percent of the thickness is used to assure that the crack growth rates shown in Figure 3 would be applicable. This provides for a ligament thickness below the crack of 0.313 inch, or a maximum crack depth of 0.937 inch. Based on this criterion, the proof test pressure should demonstrate that the minimum ligament for a long partial-thickness crack is greater than 0.313 inch with an additional allowance for crack growth. The immunity period or permissible number of pressure cycles between proof tests demonstrated by a given proof pressure, depends on the location of the crack (weld or base metal) with respect to residual stresses.

The table below shows the calculated results of a 900 psi proof test for demonstrating protection against long deep flaws, considering the effects of both 0 and 4000 psi residual stress in addition to the pressure stress acting on defects in the 36-inch piping. A proof pressure of 900 psi, for example, would cause failure of a long flaw with a 0.38 inch ligament below the flaw in a weld with 4000 psi residual stress. Similarly, the 900 psi proof pressure would cause failure of a seven-inch long crack with a 0.038 inch ligament where no residual stress is present in the pipe wall. Although the 4000 psi residual stress case provides more allowance for growth, the higher growth rates due to the higher applied stress result in fewer allowable cycles between tests. The primary reason for this is that crack growth rates are calculated by assuming the cyclic pressure range includes both the residual stress and pressure stress. Calculating ΔK for the appropriate stress level and obtaining the appropriate crack growth per cycle from Figure 5, it is possible to calculate the permissible number of cycles between proof tests as shown: (using a maximum crack depth (75% of total wall thickness) = 0.937")

<u>Residual Stress, psi</u>	<u>Demonstrated Ligament Thickness</u>	<u>Limiting Ligament Thickness</u>	<u>Available for Growth</u>	<u>Ave. da/dN, inches/1000 Cycles</u>	<u>Allowable Pressure Cycles</u>
0	0.348"	0.313	0.035"	0.0013	26,000
4000	0.381"	0.313	0.068"	0.01	6,800

Selection of Proof Pressure & Frequency

The proof pressure is based on the requirement that protection be demonstrated against the growth of hidden defects at normal and emergency operating conditions. As shown above, long, deep flaws require the highest proof pressures and are therefore limiting. The proof test pressure and frequency are based on the following data and assumptions:

- The maximum emergency overpressure noted above is 610 psi.
- Adequate protection will be provided against the accident if the overpressure does not result in a stress in the minimum ligament greater than the biaxial yield strength of the material (35,000 psi).

- The crack depth is limited to 75 percent of the wall thickness, or the depth corresponding to biaxial yield stress in the ligament below the crack at emergency conditions, whichever is larger.
- Residual stresses in the pipe welds are no higher than 4000 psi.
- Residual stresses will be considered to add to the pressure stress in calculating the stress intensity factor range to determine fatigue crack growth rates.

Evaluation of several proof pressures showed that 900 psi would provide the desired margin of safety. Test frequencies for a 900 psi proof pressure are shown in the next table for cases of 0 and 4000 psi residual stress. The ligament thicknesses corresponding to a stress in the ligament equal to biaxial yield at the emergency overpressure of 610 psi are 0.302 and 0.341 inch, respectively. Since 0.341 inch is greater than the minimum ligament required under normal conditions, this becomes the limit for the 4000 psi residual stress case.

<u>Residual Stress, psi</u>	<u>Demonstrated Ligament Thickness</u>	<u>Limiting Ligament Thickness</u>	<u>Available for Growth</u>	<u>Assumed Cyclic Stress, psi</u>	<u>Ave. da/dN, inches/1000 Cycles</u>	<u>Test Frequency Cycles</u>
0	0.348"	0.313"	0.035"	7,000	0.0013	26,000
4000	0.381"	0.341"	0.040"	11,000	0.01	4,000

The table shows that the 4000 psi residual stress case is most limiting, and would limit the reactor to 4000 pressure cycles between proof tests. However, the proof test would be repeated conservatively after 1000 cycles, to demonstrate continued protection from piping fracture due to the slow growth of hidden defects. Actually, since the K inlet piping systems were subjected to a 900 psi hydrostatic test after installation (1954), and only about 1500 pressure cycles at each K have occurred since reactor startup, it has already been shown that the systems are immune from failure by this mechanism. The temperature of the test is not crucial because the stress at tensile instability does not vary over the K reactor operating temperature range. However, the test will be performed at the low operating temperature range to also demonstrate protection against through-thickness cracks at both normal and emergency pressures.

A comparison of these critical through-thickness crack lengths for the K piping is shown in the following table: (the values are based on the combined stresses shown in Table 1 and minimum K_C values discussed above)

<u>Location</u>	<u>Critical Crack Length - Inches</u>			<u>da/dN, in/1000 Cycles (Proof-sized Crack)</u>
	<u>Proof</u>	<u>Operation</u>	<u>Emergency</u>	
36-inch riser, base metal	5.0	10.5	8.5	0.0014
36-inch riser, longitudinal weld	4.5	7.5	6.5	0.01

<u>Location</u>	<u>Critical Crack Length - Inches</u>			<u>da/dN, in/1000 Cycles (Proof- sized Crack)</u>
	<u>Proof</u>	<u>Operation</u>	<u>Emergency</u>	
36-inch horizontal pipe, base metal	5.0	10.0	8.0	0.0011
36-inch horizontal pipe, longitudinal weld	4.0	7.0	6.0	0.007
Crossheader connection to riser	3.5	6.0	4.0	0.01

The allowable crack growth before fracture, considering the maximum emergency overpressure, is 0.5 to 3.5 inches depending on location; this provides more than 20,000 cycles for the slow growth of through-the-thickness cracks to the critical size.

FUTURE PROGRAM

The fracture mechanics analysis discussed in this report will be used in conjunction with other studies to gain acceptance of the K inlet piping for reactor operation at full power. Current plans are to conduct the described proof testing in early 1970, following the completion of necessary piping modifications.

Figure 1. Fracture Specimen Locations - D Riser

TABLE 1

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SUMMARY OF NORMAL OPERATING STRESSES

	Stress Descrip.	Calc. Stress at 462 psi TORP	Vib. Press. Stress	Press. Stress Conc. Factor*	I Computed Max. Stress from Pressure		Moment-Induced Stress (T) (D) Thermal Dead Load		Moment Stress Conc. Factor**	II Computed Moment Stress (T+D) (CF)		III Residual Stress	Total Normal Operating Stress (I+II+III)
1. 1/24" Pipe	Axial	2600	300	1	2900	1200	800	--	2000	4000	8,900		
	Circum.	5800	300	1	6100	--	--	--	--	"	10,100		
2. 1/36" Pipe	A	3100	200	1	3300	1400	800	--	2200	"	9,500		
	C	6800	200	1	7000	--	--	--	--	"	11,000		
3. 1/36" Riser	A	3000	100	1	3100	Neg.	Neg.	--	--	"	7,100		
	C	6700	100	1	6800	Neg.	Neg.	--	--	"	10,800		
4. 1/36" Riser at Crossheader	A	3000	150	1	3150	Neg.	Neg.	--	--	"	7,150		
	C	6700	150	1.3	8850	Neg.	Neg.	2.0	--	"	12,850		
5. 1/36" Riser Ellipt. Head	-	--	-	-	--	--	--	--	--	"	--		
	-	6250	100	1	6350	--	--	--	--	"	10,350		
6. 1/36" Riser Hd. at Nozzle	-	--	-	-	--	--	--	--	--	"	--		
	-	6250	150	1.5	9550	--	--	--	--	"	13,550		
7. 2/36" Riser Hemi. Head	-	--	-	-	--	--	--	--	--	"	--		
	-	3150	100	1	3250	--	--	--	--	"	7,250		
8. 2/36" Riser Hemi. Head at Nozzle	-	--	-	-	--	--	--	--	--	"	--		
	-	3150	150	1.4	4550	--	--	--	--	"	8,550		
9. 1/36" Elbow Riser Base	A	3100	100	-	3200	1200	--	1.0	1200	"	8,400		
	C	6800	100	-	6900	--	--	1.0	--	"	10,900		
10. 1/36" Elbow at Tunnel	A	3100	200	-	3300	1200	800	2.2	4400	"	11,700		
	C	6600	200	-	6800	--	--	1.0	--	"	10,800		
11. 2/36x36x24 Tee at Tunnel(90°)	A	3100	200	-	3300	1200	800	2.1	4200	"	11,500		
	C	6600	200	-	6800	--	--	1.0	--	"	10,800		
12. 1/24 Ell at Tunnel	A	2600	200	-	2800	1200	800	2.2	4400	"	11,200		
	C	5800	200	-	6000	--	--	1.0	--	"	10,000		
13. 1/36x36x24 Branch Flush Conn.	A	3100	200	-	3300	Neg.	800	2.0	1600	"	8,900		
	C	6800	200	-	7000	Neg.	--	1.0	--	"	11,000		
14. 2/24x24x24 Tee 60° Lateral	A	2600	300	-	2900	1200	800	2.5	5000	***	11,900		
	C	5800	300	-	6100	--	--	1.0	--	4000	10,100		
15. 2/24x24x24 Tee	A	2600	300	-	2900	1200	800	2.1	4200	***	11,100		
	C	5800	300	-	6100	--	--	1.0	--	4000	10,100		
16. 2/24x24x12 Tee	A	2600	300	-	2900	1200	800	2.1	4200	***	11,100		
	C	5800	300	-	6100	--	--	1.0	--	4000	10,100		
17. 1/24 Ell. at Pump	A	2600	300	-	2900	--	800	2.2	1760	***	8,660		
	C	5800	300	-	6100	--	--	1.0	--	4000	10,100		

Footnotes: 1. Existing pipe and fittings
 2. New pipe and fittings as result of modifications
 3. Stresses are in pounds per square inch

* Stress concentration factors based on photoelastic models.
 ** Stress concentration factors from USAS B 31.1.0 Code for Power Piping, 1967
 *** Residual stresses in new forged fittings are actually expected to be negligible

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